ACCESSION #: 9905280183

NON-PUBLIC?: N

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Limerick Generation Station Unit 1 PAGE: 1 OF 5

DOCKET NUMBER: 05000352

TITLE: RPS and PCRVICS Actuations due to Loss of Feedwater Flow

Transient Caused by Spuriously Opening Breaker

EVENT DATE: 04/20/1999 LER #: 1999-003-00 REPORT DATE: 05/19/1999

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

Other

LICENSEE CONTACT FOR THIS LER:

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Manager-Experience Assessment

COMPONENT FAILURE DESCRIPTION:

CAUSE: Other SYSTEM: EC COMPONENT: BRK MANUFACTURER: C770

Other AD BRK I202

REPORTABLE EPIX: Y

Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On April 20, 1999, at 1830 hours, Limerick Generating Station (LGS) Unit 1 was operating at 100% power when a loss of feedwater was experienced. The Main Control Room received a condensate filter trouble, a Reactor Low Level, and Loss of Feedwater alarms. All three Reactor Feed Pumps tripped on low suction pressure causing a reactor SCRAM when level decreased to + 12.5 inches. The Reactor Protection System actuation is reportable pursuant to 10CFR50.73(a)(2)(iv).

The low reactor water level Reactor Protection System actuation occurred due to a spurious tripping of a breaker that supplies position indication to the inlet valves of the Deep Bed Condensate Demineralizer System (DBCDS). Loss of inlet position indication to the DBCDS programmable logic controller created a signal for the deep bed outlet valves to close. The resultant rapid increase in differential pressure at the DBCDS bypass valves prevented their ability to be opened and maintain flow to the Reactor Feed Pumps and the reactor. This event is bounded by the previously analyzed loss of feedwater flow transient as described in the LGS Updated Final Safety Analysis Report (UFSAR). The plant responded as designed to a loss of feedwater flow. The DBCDS breaker that spuriously tripped was replaced and the DBCDS breakers for the outlet valves were procedurally opened to prevent closure.

This event resulted in High Pressure Core Injection, and Reactor Core Isolation Cooling injections into the Reactor Coolant System. SPECIAL REPORT information required pursuant to Technical Specification (TS) 6.9.2 and TS Limiting Conditions for Operation ACTION statements 3.5.1.f and 3.7.3.b is included in this report.

EIIS Codes are in []

Unit Conditions Prior to the Event

Unit 1 was in Operational Condition (OPCON) 1 (Power Operation) at 100%

power.

Description of the Event

On April 20, 1999 at 1830 hours, a breaker [EC] supplying power to the deep

bed demineralizer [SF] inlet valve position indication to a programmable

logic controller (PLC), spuriously tripped open. The PLC, sensing a loss

of inlet valve position indication, is programmed to close the motorized

deep bed demineralizer outlet valves. The resultant increasing differential pressure across the deep bed demineralizers signalled the deep bed demineralizer bypass valves to open. The rapid increase of differential pressure around the demineralizers exceeded the ability of the 30" and 12" bypass valves to open, causing them to each trip on their associated thermal overloads. The resultant loss of feedwater flow caused all three Reactor Feed Pumps [SK] to trip on low suction pressure and a decreasing reactor pressure vessel (RPV) [AD] level. When the level decreased to +12.5", the Reactor Protection System (RPS) [JC] automatically scrammed the reactor. All control rods fully inserted. In addition to an automatic reactor scram, the appropriate automatic Primary Containment and Reactor Vessel Isolation Control System (PCRVICS) isolations, an Engineered Safety Feature (ESF), occurred at low RPV water level. The reactor level continued to drop to the -38" level resulting in the initiation of additional PCRVICS isolations and actuation of the High Pressure Coolant Injection (HPCI), an Emergency Core Cooling System (ECCS), and Reactor Core Isolation Cooling (RCIC) Systems which injected into the reactor vessel.

In response to Redundant Reactivity Control System (RRCS) - Alternate Rod Insertion (ARI) initiation, the Anticipated Transient Without Scram (ATWS) - Recirc Pump Trip (RPT) breakers tripped both Recirculation Pumps, however, one of the two in-series breakers on the 'A' Recirc pump failed to trip.

Reactor level continued to decrease to a level of -74" (lowest level for the event). Top of active fuel is -161 ". HPCI and RCIC systems recovered reactor level to +25". The Main Turbine was manually tripped after a transfer of house loads. Condenser vacuum was lost due to the loss of feedwater flow to the Steam Jet Air Ejector Condenser. Auxiliary steam was placed in service to the steam seal evaporator to maintain turbine steam seals. The mechanical vacuum pump was utilized to recover condenser vacuum. HPCI and RCIC were used for level and pressure control until the feedwater system was restored.

A one hour notification was made to the NRC at 1921 hours on April 20, 1999, in accordance with the requirements of 10CFR50.72(b)(1)(iv) since this event resulted in an ECCS discharge into the reactor coolant system as a result of a valid signal. This also satisfied the four hour notification requirement of 10CFR50.72(b)(2)(ii) for the automatic RPS and ESF actuations. This LER is being submitted in accordance with the requirements of 10 CFR50.73(a)(2)(iv).

Analysis of Event

The safety significance of this event was minimal. This event is bounded by the previously analyzed loss of feedwater flow transient as described in the Limerick Generating Station Updated Final Safety Analysis Report (UFSAR). The plant responded as designed to a loss of feedwater flow. Adequate core cooling was maintained and RPV pressure was controlled throughout the event. Additionally, no safety relief valves [SB] were actuated and there was no challenge to the primary containment [NH].

The Deep Bed Condensate Demineralizer System (DBCDS) bypass valves were not designed for a scenario of a fast, large differential pressure transient such as occurred during this event. The results were the tripping of the bypass valve motor operators on thermal overload.

The Main Steam Relief Valves (MSRVs) - Automatic Drepressurization System (ADS) and Low Pressure ECCS were also available to mitigate this event.

Cause of Event

The cause of this event was spurious tripping of the breaker which supplies power to the deep bed demineralizer inlet valve position indication. The DBCDS control logic has a failure mode which causes all of the vessel outlet flow control valves to close simultaneously. The failure mode involves the loss of power to the position indication circuits for the auxiliary (aux) valves and the vessel inlet (A) valves which provide input signals to the control logic in the programmable logic controller (PLC). The logic for the vessel outlet flow control valves includes a permissive which requires all auxiliary valves to be full closed and the vessel inlet valve to be full open. Upon an incorrect position OR loss of position indication signals for any of these valves, the corresponding deep bed demineralizer outlet (E) valve will receive a signal to close. This signal overrides all other position setting signals to the E valve of each vessel and is in effect at all times. The simultaneous loss of position indication signal to all eight deep bed demineralizer inlet valves causes all of the vessel outlet valves to close simultaneously. However, the close signal will only be sent to the outlet valves if the PLC loses the position indication input without losing power from its own regulated power source. If both power supplies are lost, the outlet valves would remain in the open position.

A review was conducted to evaluate the potential causes of the spurious breaker tripping. There were no known disturbances on the incoming 33kV offsite Moser power supply fine during the event. Onsite outage activities involving power usage on turbine deck and condenser areas were found to be within reason (no unusual power usage or fluctuations). The most probable causes of the breaker trip appear to be either a spurious trip or an anomaly in power quality on the circuit. The breaker was replaced to address the first cause. Post-event testing by Maintenance did not indicate any problems in the breakers removed from the field. The breaker that spuriously tripped was sent for testing at Valley Forge Corporate Labs. Analysis of the breaker revealed no abnormalities that would lead to a cause for spurious tripping. The breaker is a Cutler-Hammer/ Eaton, Model FH3600, 600 volt molded case circuit breaker. Industry review of data revealed no similar events of spurious opening of FH3600 breakers. Operating experience review revealed several instances of spurious opening of Cutler-Hammer breakers.

Initial visual investigation of the 1A ATWS-RPT breaker failure identified a damaged trip coil #1. The coil showed signs of overheating. The plunger was frozen in place and the coil resistance was low. The breaker in question is an ABB, 5HK250 breaker which is supplied with two independent trip coils. Inspection of all other mechanical components in the trip mechanism showed no indications of grease hardening. Corporate Labs testing of the coils indicated that the failure mode of a test coil clamped/stalled at 120 VDC appears similar in nature to the trip coil #1 found failed in the 1A ATWS-RPT. The failure of the breaker to trip was caused by a failure of trip coil #1 , or mechanical resistance which prevented the trip coil from opening the breaker and subjecting it to an extended exposure at rated voltage and subsequent overheating.

Corrective Actions

The circuit breaker that spuriously tripped was replaced with a new circuit breaker. The 1A ATWS-RPT breaker was also replaced with a new breaker.

The DBCDS system operating procedures were changed to open the breakers for the vessel outlet flow control valves after placing the system in service.

This corrective action will ensure that the DBCDS PLC logic will be enabled as the vessels are placed in service (the intended function) but opening the breakers will ensure the valves remain open to provide a feedwater flow path in the event of the loss of non-regulated power. This action will be completed on Unit 2 prior to restart from the current refuel outage.

Data collection to monitor power quality of the offsite power feed to the breaker has been completed. Analysis reveals no significant abnormalities.

Previous Similar Occurrences

None.

SPECIAL REPORT

This Special Report is being submitted pursuant to the requirements of Limerick Generating Station Technical Specification 3.5.1.f, 3.7.3.b and 6.9.2. Specification 3.5.1.f states, "In the event an ECCS system is actuated and injects water into the reactor coolant system, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70". Specification 3.7.3.b states, "In the event the RCIC system is actuated and injects water into the reactor coolant system, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date." Technical Specification 6.9.2 states, "Special Reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report".

This report concerns the occurrence of one High Pressure Coolant Injection (HPCI) System Actuation and four Reactor Core Isolation Cooling (RCIC) system actuations and injections into the reactor coolant system of Unit No. 1.

Below is a description of each of the HPCI and RCIC system actuation and injection events.

On April 20, 1999 the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems actuated and injected to the reactor pressure vessel. The actuation occurred automatically in response to reactor vessel low water level following a reactor scram caused by loss of feedwater. Reactor parameters prior to the transient were as follows:

Reactor Power - 3455 MWt

Reactor Coolant System Pressure 1049.7 PSIG

Monitor Temperature 531.7 degrees F

Core Flow 89.51 Mlbm/hr

Feedwater Flow 15-04 Mlbm/hr

Feedwater Temperature 430.3 degrees F

Average HPCI flow was approximately 5,600 gpm which operated continuously for approximately 8 and one-half minutes. This constitutes the fifth HPCI actuation and injection to date, for the life of the plant. The current usage factor for the safety injection nozzles for the HPCI injection is less than 0.70.

During this transient, RCIC was used for level control. The initial actuation and injection was automatic as a result of low reactor water level. The following three injections were manually actuated and injected as required to maintain level during reactor cool down. Total injection time was approximately 1 6 hours. These constitute the 20th, 21st, 22nd and 23rd RCIC actuations and injections to date, for the life of the plant. The flow rate and duration of the first RCIC injection were approximately 600 gpm and 3 hours and 17 minutes, respectively. The second RCIC injection was approximately 2 hours and 46 minutes in length. The average RCIC flow during the second injection was approximately 600 gpm. Reactor parameters prior to the second RCIC injection were as follows:

Reactor Coolant System Pressure 925 PSIG

Monitor Temperature 497 degrees F

Core Flow 0.0 Mlbm/hr

Feedwater Flow 0.0 Mlbm/hr

The third RCIC injection was approximately 27 minutes in length. The average RCIC flow during the third injection was 41 2 gpm. Reactor parameters prior to the third RCIC injection were as follows:

Reactor Coolant System Pressure 471 PSIG

Monitor Temperature 466 degrees F

Core Flow 0.0 Mlbm/hr

Feedwater Flow 0.0 Mlbm/hr

The fourth RCIC injection was approximately 8 hours and 26 minutes in length. The average RCIC flow during the fourth injection was approximately 472 gpm. Reactor parameters prior to the fourth RCIC injection were as follows:

Reactor Coolant System Pressure 421 PSIG

Monitor Temperature 466 degrees F

Core Flow 0.0 Mlbm/hr

Feedwater Flow 0.0 Mlbm/hr

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